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Your ref: Docket No. 52-006
Our ref: DCP/NRC1635

October 10, 2003

SUBJECT: Transmittal of Revised Responses to AP1000 DSER Open Items

This letter transmits Westinghouse revised responses to Open Items in the AP1000 Design Safety Evaluation Report (DSER). A list of the revised DSER Open Item responses transmitted with this letter is Attachment 1. The non-proprietary responses are transmitted as Attachment 2.

Please contact me at 412-374-5355 if you have any questions concerning this submittal.

Very truly yours,

M. M. Corletti

M. M. Corletti
Passive Plant Projects & Development
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/Attachments

1. List of the AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses transmitted with letter DCP/NRC1635
2. Non-Proprietary AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Responses dated October 10, 2003

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October 10, 2003

Attachment 1

List of

Non-Proprietary Responses

Table 1 "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1635"	
2.5.4-2 Revision 2 3.7.2.3-2 Revision 1 3.7.2.9-1 Revision 1 3.8.3.5-3 Revision 1 5.3.3-1 Revision 1 19A.2-7 Revision 2 19A.2-9 Revision 2	

October 10, 2003

Attachment 2

AP1000 Design Certification Review

Draft Safety Evaluation Report Open Item Responses

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **2.5.4-2 (Revision 2)**

Original RAI Number(s): **None**

Summary of Issue:

The bearing capacity of the subgrade is a fundamental design parameter for this standard design. In the design of the foundation of a large structure it is important to ensure that under normal operating conditions, the average pressure on the subgrade is less than the allowable average bearing capacity of the foundation material, and that the peak subgrade pressure caused by the load combination with the SSE imposing the largest toe pressure at the edge of the foundation is also within the allowable capacity of the subgrade. The allowable bearing capacity of the subgrade is governed by settlement or crushing. Under relatively soft soil conditions, short term soil movement due to water table fluctuation and long term settlement due to the super imposed loading affect the allowable bearing capacity. Under hard rock subgrade conditions, the bedding direction of rock layers and the level of cracking and other discontinuities in the matrix of the rock material can limit the allowable average and allowable peak bearing capacity. The response to the RAIs indicates that the bearing capacity at a hard rock site will exceed 21.55MPa (450,000 pounds per square ft). During the April 2 through 5, 2003 audit, the staff requested the applicant to clearly specify, in the DCD, that this standard design is based on an allowable average and an allowable peak bearing capacity, and should specify what these values are. This is Open Item 2.5.4-2.

In a telephone discussion on August 22, 2003 **and subsequent discussions in the meeting on October 6-7, 2003**, it was requested that additional information be provided on the bearing demand and the determination of the allowable bearing capacity by the Combined License applicant.

Westinghouse Response (Revision 2~~1~~):

This Open Item was addressed by changes to Chapter 2 included in DCD Revision 5. The changes were made in the response to RAI 240.005 transmitted by letter DCP/NRC1586 on May 7, 2003.

A revision to the DCD is shown below to address the additional comments from the NRC in the August 22 telephone call **and the October 6-7, 2003 meeting**. The revisions also include a correction to the required bearing capacity (120,000 pounds per square foot) based on the latest revision of the nuclear island basemat analyses. This bearing demand replaces the demand of 450,000 pounds per square ft. which was given in the earlier revision of the DCD prior to completion of the basemat analyses.

Typical allowable bearing capacities for rock were provided in the revision 1 response to RAI 241.001. Allowable bearing pressures for rock given in Reference 2.5.4-2-1 range from 80 ksf to in excess of 200 ksf (Boston, Denver, Newark, New York, Philadelphia, and New York city).

It is recognized that it is difficult to establish a generic methodology to determine the allowable bearing capacity since this is site dependent. As stated by Terzaghi and Peck (Reference 2.5.4-2-1): "Because of the great variety of soils and combinations of soils encountered in practice, no single method for determining the allowable soil pressure can be developed that would be suitable under all circumstances. The procedure must always be adapted to the soil conditions revealed by the exploratory borings. ..." Therefore, the COL applicant is referred to the acceptance criteria given in the Standard Review Plan 2.5.4 in the DCD.

References:

2.5.4-2-1 Terzaghi, Karl, and Ralph B. Peck, Soil Mechanics in Engineering Practice, John Wiley & Sons, Inc., New York, 1948, p 418.

Design Control Document (DCD) Revision:

Revise soil bearing parameters in Table 5.0-1 of Tier 1 and Table 2-1 of Tier 2 as follows:

Soil

Average Allowable Static Bearing Capac~~bil~~ity	Greater than or equal to 8,600 lb/ft^2 over the footprint of the nuclear island at its excavation depth
Maximum Allowable Dynamic Bearing Capacity for Normal Plus SSE	Greater than or equal to ~~85~~120,000 lb/ft^2 at the edge of the nuclear island at its excavation depth
Shear Wave Velocity	Greater than or equal to 8,000 ft/sec based on low-strain best-estimate soil properties over the footprint of the nuclear island at its excavation depth
Liquefaction Potential	None

Revise subsection 2.5.4.2 as shown in DCD Revision 7 as follows:

2.5.4.2 Bearing Capacity

The maximum bearing reaction on the hard rock determined from the analyses described in subsection 3.8.5.1 is less than ~~85,~~120,000 pounds per square foot under all combined loads including the safe shutdown earthquake. **The allowable** ~~b~~Bearing capacity at a hard rock site will exceed this demand.

The maximum bearing reaction on the hard rock specified in Table 2-1 is determined from the analyses described in subsection 3.8.5.1. ~~These analyses consider the foundation as a very stiff semi-infinite elastic medium. This results in high bearing reactions below the stiff walls of the nuclear island. Where the rock is unable to support these high local bearing pressures, loads will redistribute to the adjacent rock. The key attribute for acceptability of the site for an AP1000 is the structural capacity of the mat to resist the bearing pressure. The mat has substantial margin to accommodate a redistribution of the bearing reactions. Evaluation criteria are defined to evaluate sites that do not satisfy the site parameters directly.~~ **The evaluation of the allowable capacity of the bedrock should be based on the properties of the underlying materials (see subsection 2.5.4.5.2) including appropriate laboratory test data to evaluate strength, and considering local site effects, such as fracture spacing, variability in properties, and evidence of shear zones. The allowable bearing capacity should provide a factor of safety appropriate for the design load combination including safe shutdown earthquake loads.**

If the shear wave velocity or the allowable bearing capacity are outside the range evaluated for AP1000 design certification, a site specific evaluation can be performed using the AP1000 basemat model and methodology described in subsection 3.8.5. The safe shutdown earthquake loads are those from the AP1000 analyses described therein. Alternatively, bearing pressures may be determined from a site-specific analysis using site specific inputs as described in subsection 2.5.2.3. For the site to be acceptable the bearing pressures from the site-specific analyses including static and dynamic loads need to be less than the capacity of each portion of the basemat.

Revise subsection 2.5.4. 5.6 as shown in DCD Revision 7 as follows:

2.5.4.5.6 Bearing Capacity – The Combined License applicant will verify that the site-specific allowable soil bearing capacities for static and dynamic loads are equal to or greater than the values documented in Table 2-1 or will provide a site specific evaluation as described in subsection 2.5.4.2. **The acceptance criteria for this evaluation are those of Standard Review Plan 2.5.4 as follows:**

- **The static and dynamic loads, and the stresses and strains induced in the soil surrounding and underlying the nuclear island are conservatively and realistically evaluated**
- **The consequences of the induced soil stresses and strains, as they influence the soil surrounding and underlying the nuclear island have been conservatively assessed.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.7.2.3-2 (Revision 1)

Original RAI Number(s): 230.018

Summary of Issue:

In its response to RAI 230.18 Response Revision 3 transmitted by Westinghouse letter DCP/NRC1588, dated May 13, 2003, the studied the studied the effect of the steel containment vessel vertical response by the use of a multi-mass model of the polar crane instead of the single mass model used in the AP600 analyses and the initial AP1000 analyses. The staff would like Westinghouse to cite any other reasons that can explain the change in the steel containment vessel vertical response.

Westinghouse Response (Revision 1):

The maximum vertical absolute acceleration of the steel containment vessel is 1.49g for the AP600. In the original DCD the AP600 stick model was "stretched" to match the AP1000 height dimensions, and maximum vertical absolute acceleration of the steel containment vessel became 1.40g for the AP1000. In the most recent AP1000 analyses shown in DCD Revision 7, the maximum acceleration is 1.25g.

The reduction in vertical response is associated with two types of changes that were done to the stick model. The first type of changes applies to the Auxiliary/Shield Building (ASB). The ASB stick model properties are no longer based on "stretched" AP600 dimensions, but are now based finite element analyses which used the actual AP1000 dimensions to calculate stiffness and the develop more realistic AP1000 ASB stick model dimensions. The second type of change is that the model now includes better definition of the AP1000 polar crane and the use of a multi-mass model of the polar crane instead of the single mass used in the AP600 analyses and the initial AP1000 analyses. The maximum vertical absolute acceleration of the steel containment vessel after the two changes described above were implemented became 1.13g for the AP1000.

Based on a number of studies performed and submitted earlier, it is evident that the changes to the Auxiliary Building stick model properties is the major reason why the spectra values are reduced. Changing the polar crane to a multi-mass model also is a lesser effect that contributes to the reduction in maximum vertical absolute acceleration of the steel containment vessel.

In the spectra presented in DCD Rev. 7, two additional changes were applied to the model. First was the concrete stiffness was reduced by applying a 80%E factor to all the concrete elements. Second, the steel containment vessel (SCV) was directly connected to the CIS stick rather than to ASB stick. These effects increased the maximum acceleration 1.25g for the AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.7.2.9-1 (Revision 1)

Original RAI Number(s): 230.020

Summary of Issue:

As described in DCD Tier 2 Section 3.7.2.9, as part of accounting for parameter uncertainty, Westinghouse in its response to DSER Open Item 3.7.2.3-1 shows floor response spectra which include the effects of stiffness reduction due to shear wall concrete cracking. In comparing the floor response spectra published in DCD Revision 3 which was based on uncracked concrete properties (100% E) , and DCD Revision 6 which is based on partially cracked concrete properties (80% E), it is found that in most instances the 80% E analysis, shows increases in floor response spectra peaks. There are a few instances however where the floor spectra peaks have been reduced. Examples of this are the floor response spectra in the EW (Y) direction of the containment internal structures nodes 532 and 535. The staff would like Westinghouse to cite the reasons that can explain the change in the reduction at these two locations.

Westinghouse Response (Revision 1):

Figures 3.7.2.9-1-1 and 3.7.2.9-1-2 show the floor response spectra in the EW (Y) direction of the containment internal structures nodes 532 and 535. The differences between the Rev 2 100% E spectra and the Rev 3 100 %E spectra are due to a more accurate representation of the nuclear island model (including the SCV being attached to the CIS). This more accurate representation included smaller masses at these locations as well as new beam properties based on the CIS finite element model stiffness analyses. The differences between the Rev 3 100% E spectra and the Rev 3 80 %E spectra are due to change from uncracked to partially cracked concrete properties. The reduced stiffness changes cause slight changes in the nuclear island mode shapes that reduce the response in this location.

This response also addresses issues raised in the Structural Audit held at Westinghouse offices on October 6 –9, 2003 related to DSER Open Item 3.7.2.3-1.

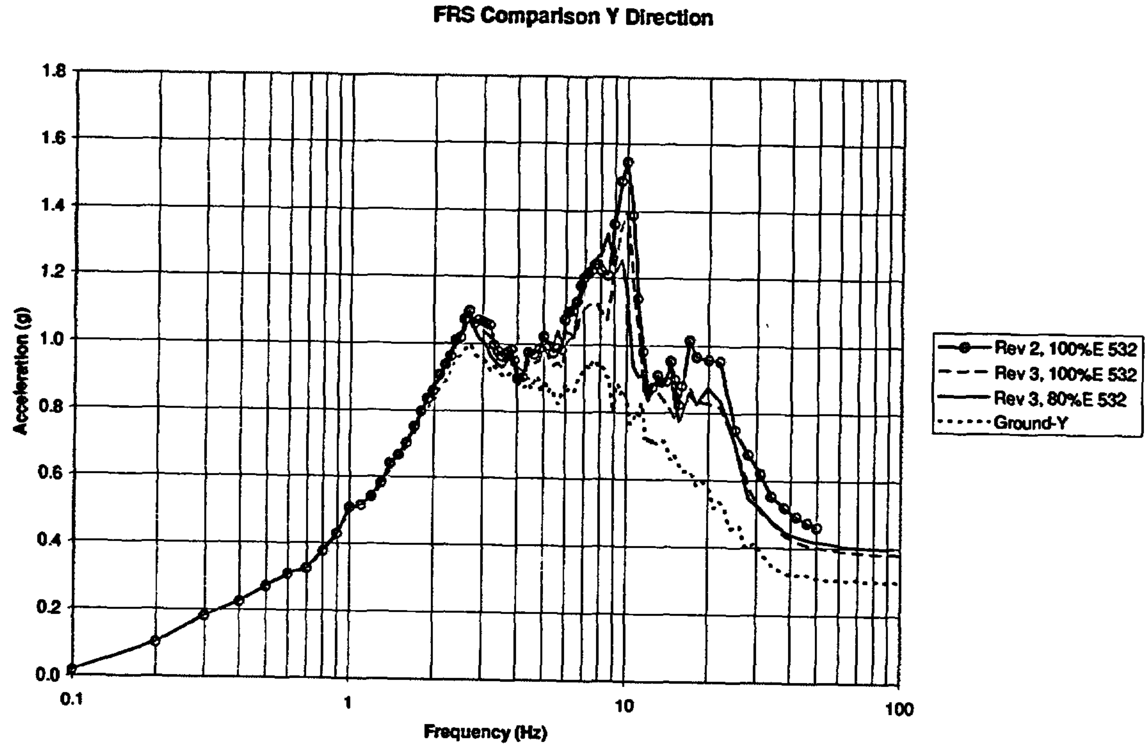


Figure 3.7.2.9-1-1: Node 532 – FRS comparison - Y Direction

<u>Note</u>: Rev. 2 was included as part of DCD Rev 3
Rev. 3 was included as part of DCD Rev 6

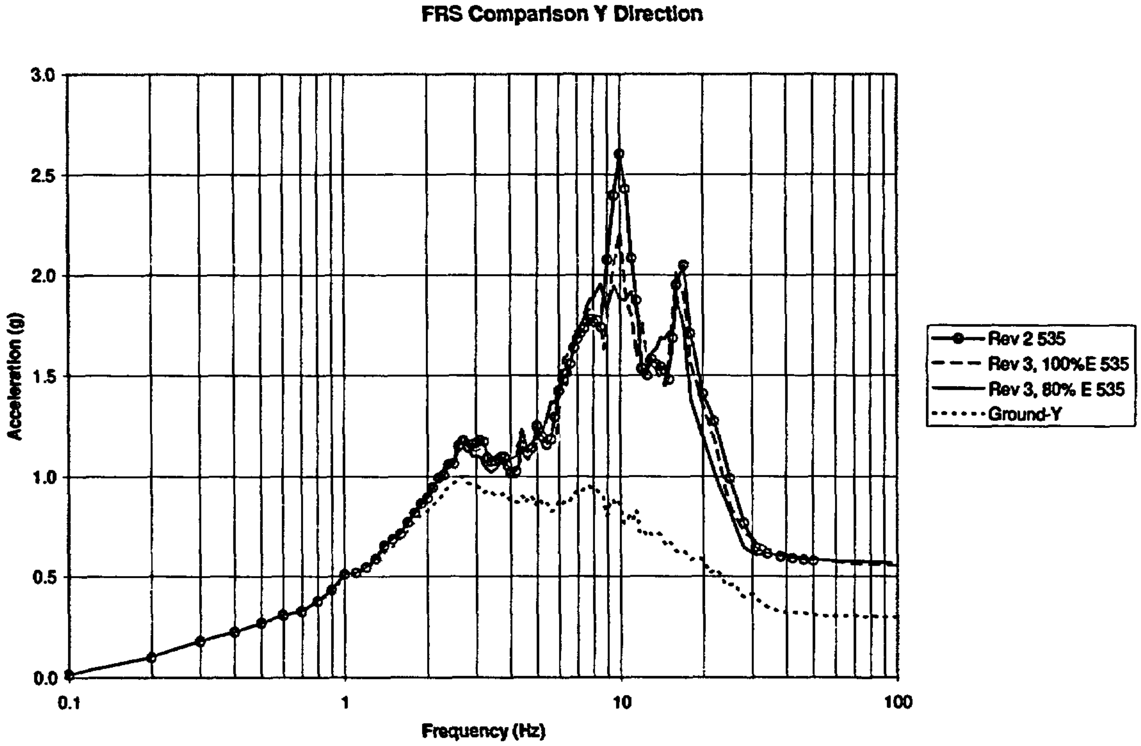


Figure 3.7.2.9-1-2: Node 532 – FRS comparison - Y Direction

<u>Note</u>: Rev. 2 was included as part of DCD Rev 3
Rev. 3 was included as part of DCD Rev 6

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 3.8.3.5-3 (Revision 1)

Original RAI Number(s): None

Summary of Issue:

Westinghouse calculation No. APP-1100-S2C-007, Revision 0, contains the design of the IRWST concrete-filled steel module walls. The staff reviewed the approach used to calculate the needed steel area of the structural walls. The calculation determined the necessary steel reinforcement area at various locations in each of the critical walls. This was done using the methodology contained in Westinghouse guidance document APP-GW-S1-008, Revision 0 (Design Guide for Reinforcement in Walls and Floor Slabs). During the audit, the applicant indicated that boundary elements are not needed for walls that frame into other walls since the other walls act as boundary elements. The staff found that the applicant's approach for the analysis and design does not meet the criteria of Chapter 21.6, "Structural Walls, Diaphrams and Trusses," of ACI-349-01 in which, the criteria for using boundary elements are specified. A similar issue is presented in Subsection 3.8.4.2 of this report under Open Item 3.8.4.2-1. This is Open Item 3.8.3.5-3.

Additional information on the stress levels was discussed during the meeting on October 6 – 10 and is included in the Revision 1 response.

Westinghouse Response (Revision 1):

The open item on boundary elements for reinforced concrete walls is addressed in the response to DSER Open Item 3.8.4.2-1. Inside containment the walls are constructed using concrete filled steel modules. Typical corner details for these modules are shown in DCD Figure 3.8.3-8, Sheet 1. The steel plates provide excellent confinement for the concrete and the stiffeners limit potential buckling. The thickness of walls is established by shielding and constructibility considerations and the stresses are low. The corner details and low stresses result in a design satisfying the intent of Chapter 21.6 of ACI-349-01.

The design of the critical sections is summarized in DCD subsection 3.8.3.5.8. The maximum compressive stress in the module walls occurs at the location summarized in sheet 3 of Table 3.8.3-5. At this location the compressive axial member force (TY) under normal operating loads plus the safe shutdown earthquake ($D + F + L_o + E_s$) is 273 k/ft giving a stress in the concrete of 616 psi. This is less than the threshold of 0.2 f_c' specified in ACI 349-01. There are no openings in these walls. Thus, boundary elements are not required for the critical sections inside containment.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **5.3.3-1 (Revision 1 Response)**

Original RAI Number(s): **251.018**

Summary of Issue:

The staff requested, in RAI 251.018, that the applicant demonstrate that the P-T limits are in accordance with Appendix G to 10 CFR Part 50. The applicant responded, that the AP1000 heatup and cooldown operating curves were generated using the most limiting adjusted reference temperature values and the NRC-approved methodology as documented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with staff approved exceptions.

One exception is that instead of using best estimate fluence values, the applicant is using fluence values that are calculated fluence values. The staff finds this acceptable because this is in compliance with RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The other exception is that the KIc critical stress intensities are used in place of the KIa critical stress intensities. This methodology is taken from staff approved ASME Code Case N-641. The staff found the applicant's responses acceptable because the AP1000 P-T limit curves were developed in accordance with 10 CFR Part 50, Appendix G, with the exception that the flange requirement is in accordance with WCAP 15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." Currently, the staff has not approved WCAP 15315. Any changes to the RV closure head requirements would be incorporated into Appendix G of 10 CFR Part 50. If a relaxation to 10 CFR Part 50, Appendix G is approved, this will allow the operating window to be wider. Since applicants using AP1000 are required to meet the requirements of 10 CFR Part 50, Appendix G, applicants using AP1000 must meet the closure head requirements of Appendix G of 10 CFR Part 50. However, the AP1000 DCD does not provide limitations (values of RTNDT) for the closure flange region of the RV and head. The AP1000 design must include these limitations in order to satisfy Appendix G of 10 CFR Part 50. The applicant should provide these limitations that are consistent with the present TSs and 10 CFR Part 50, Appendix G, or provide closure flange limitations with new TSs that are consistent with 10 CFR Part 50, Appendix G. This is Open Item 5.3.3-1.

Westinghouse Response (Revision 1):

~~Since it is recognized that the elimination of the flange requirement, as discussed in WCAP 15315, results in plant safety and operational improvements, Westinghouse proposes to maintain the P/T curves without the flange requirement in the AP1000 DCD and request exemption from the 10 CFR Part 50 Appendix G flange limits. Westinghouse requests further interaction with the NRC staff to resolve any technical issues associated with this exemption.~~

~~When evaluating the request for exemption, consideration should be given to the COL item in DCD Section 5.3.6.1 in which it is recognized that the P/T curves given in the DCD are generic~~

~~curves and that the Combined License Applicant is committed to addressing P/T curves based on the as-procured reactor vessel material. An LTOPS evaluation, including assessment of the RHR relief valve setpoint and relief capacity, is also committed to be performed to determine the impact of any changes in the P/T curves.~~
Westinghouse will revise the AP1000 DCD to include P/T curves that meet the reactor vessel closure head flange requirements of 10 CFR Part 50 Appendix G. The normal RHR system relief valve setpoint and capacity will also be revised as a result of a revised LTOPS evaluation based on the new P/T curves.

The impact of the revised normal RHR system relief valve setpoint and capacity on the analyses of a loss of normal RHR cooling in Mode 4 with the RCS intact (DCD Section 19E.4.8.2) is being evaluated.

A review of the ITAAC associated with the normal RHR system relief valve (Tier 1 Section 2.3.6) shows that specification of the relief valve capacity based on the generic P/T curves in the DCD is inconsistent with the COL item in Section 5.3.6.1. The COL item requires an evaluation of the adequacy of the normal RHR system relief valve based on the P/T curves developed for the as-procured reactor vessel material, which could result in a revised required relief valve capacity. The ITAAC associated with the normal RHR system relief valve will be revised to a more general requirement so that this ITAAC is compatible with the possibility of changes in the required capacity of the valve as a result of P/T curves based on as-procured reactor vessel material.

Design Control Document (DCD) Revision (From Revision 0 Response – Incorporated Into DCD Revision 7):

From DCD Tier 1, Section 2.3.6, Table 2.3.6-4, page 2.3.6-12:

Table 2.3.6-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.a) The Class 1E equipment identified in Tables 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	A report exists and concludes that the Class 1E equipment identified in Table 2.3.6-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

7.b) The Class 1E components identified in Table 2.3.6-1 are powered from their respective Class 1E division.	Testing will be performed on the RNS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.3.6-1 when the assigned Class 1E division is provided the test signal.
7.c) Separation is provided between RNS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.
8.a) The RNS preserves containment integrity by isolation of the RNS lines penetrating the containment.	See Tier 1 Material, subsection 2.2.1, Containment System.	See Tier 1 Material, subsection 2.2.1, Containment System.
8.b) The RNS provides a flow path for long-term, post-accident makeup to the RCS.	See item 1 in this table.	See item 1 in this table.
9.a) The RNS provides LTOP for the RCS during shutdown operations.	i) Inspections will be conducted on the low temperature overpressure protection relief valve to confirm that the capacity of the vendor code plate rating is greater than or equal to system relief requirements. ii) Testing and analysis in accordance with the ASME Code Section III will be performed to determine set pressure.	i) The rated capacity recorded on the valve vendor code plate is not less than ~~650 gpm~~**the flow required to provide low-temperature overpressure protection for the RCS, as determined by the LTOPS evaluation based on the P/T curves developed for the as-procured reactor vessel material.** ii) A report exists and concludes that the relief valve opens at a pressure such that the relief capacity is not less than ~~650 gpm at a pressure of 900 psig.~~**the flow required to provide low-temperature overpressure protection for the RCS.**

Design Control Document (DCD) Revisions (From Revision 1 Response):

From DCD Revision 7, page 1.6-12, Table 1.6-6

Table 1.6-1 (Sheet 11 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
5.2	WCAP-8324-A	Control of Delta Ferrite in Austenitic Stainless Steel Weldments, June 1975
	WCAP-8693	Delta Ferrite in Production Austenitic Stainless Steel Weldments, January 1976
5.3	WCAP-15557	Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology, August 2000
	WCAP-14040-NP-A	**Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves**
5.4	WCAP-15994-P (P) WCAP-15994-NP	Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel, March 2003

From DCD Revision 7, page 5.2-7, Section 5.2.2.2:

Subsection 5.4.9 discusses the capacities of the pressurizer safety valves and residual heat removal system relief valve used for low temperature overpressure protection. The setpoints and reactor trip signals which occur during operational overpressure transients are discussed in subsection 5.4.5. With the current AP1000 pressure-temperature limits (subsection 5.3.3), the set pressure for the relief valve in the normal residual heat removal system is based on a sizing analysis performed to prevent the reactor coolant system pressure from exceeding ~~110 percent of the design pressure of the normal residual heat removal system~~**the applicable low temperature pressure limit for the reactor vessel based on ASME Code, Section III, Appendix G.** The limiting mass and energy input transients are assumed for the sizing analysis.

From DCD Revision 7, page 5.3-13, Section 5.3.3.1:

The pressure-temperature curves are developed considering a radiation embrittlement of up to 54 effective full power years (EFPY) consistent with an expected plant design life of 60 years with

90 percent availability. Copper, nickel contents and initial RT_{NDT} for materials in the reactor vessel beltline region and the reactor vessel flange and the closure head flange region are shown in Tables 5.3-1 and 5.3-3. The operating curves are developed with the methodology given in Reference 6, which is in accordance with 10 CFR 50, Appendix G with the following exceptions:

1. The fluence values used are calculated fluence values (i.e., comply with Regulatory Guide 1.190), not the best-estimate fluence values.
2. The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641 (which covers Code Cases N-640 and N-588).
3. The 1996 Version of Appendix G to Section XI is used rather than the 1989 version.

~~4.The flange requirement is not considered per Reference 7.~~

From DCD Revision 7, page 5.3-23:

5.3.7 References

1. ASTM E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."
2. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," United States Nuclear Regulatory Commission, Office of Nuclear Reactor Research, March, 2001.
3. WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," S. L. Anderson, August 2000.
4. NRC Policy Issue, "Pressurized Thermal Shock," SECY-82-465, November 23, 1982.
5. Theofanous, T.G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.
6. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.

~~7.WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," W. Bamford, et al., October 1999.~~

From DCD Revision 7, page 5.3-32:

Current Figure 5.3-2

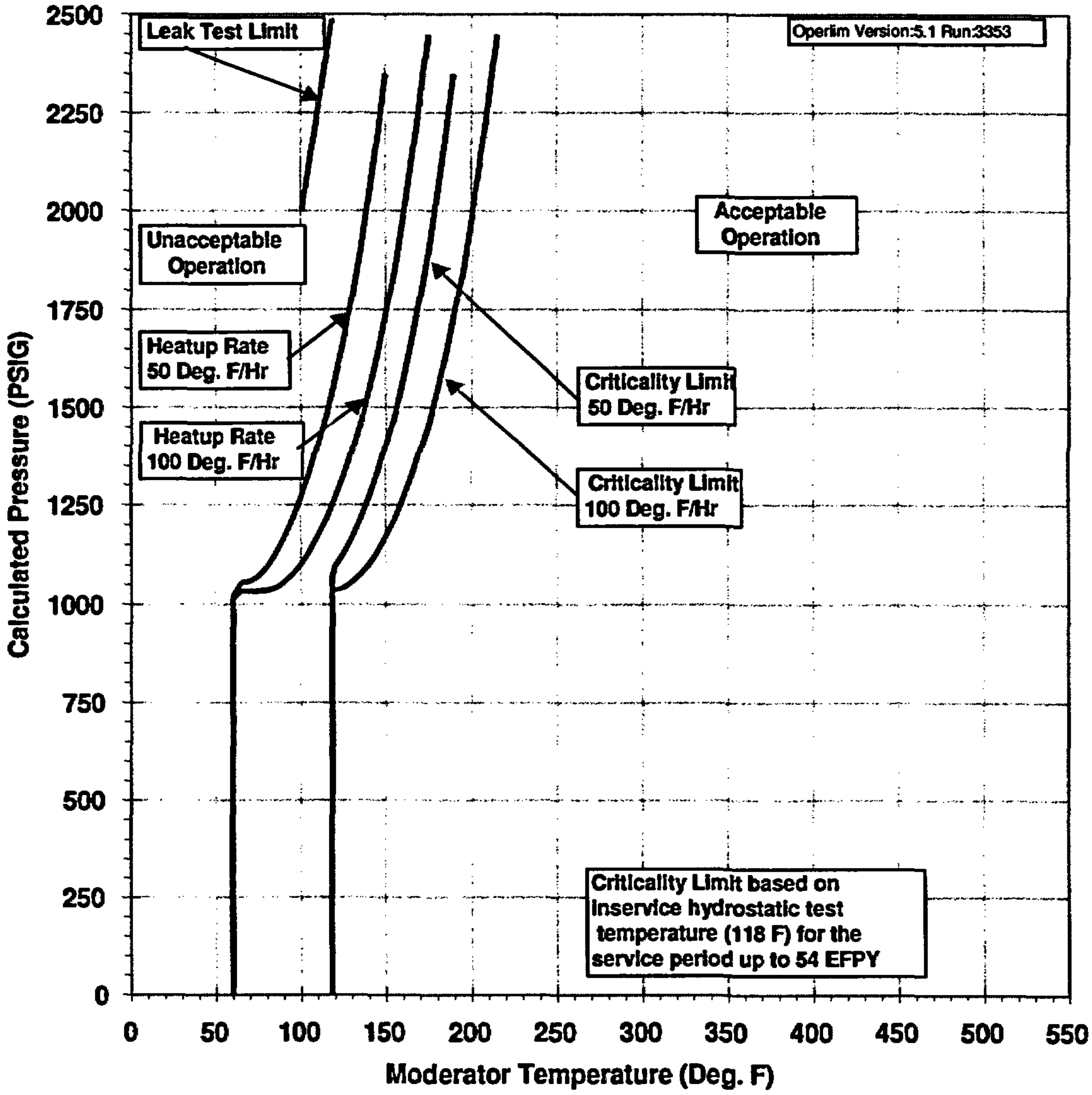


Figure 5.3-2

AP1000 Reactor Coolant System Heatup Limitations (Heatup Rate Up to 50 and 100°F/hour) Representative for the First 54 EFPY (Without Margins for Instrumentation Errors)

Revised Figure 5.3-2

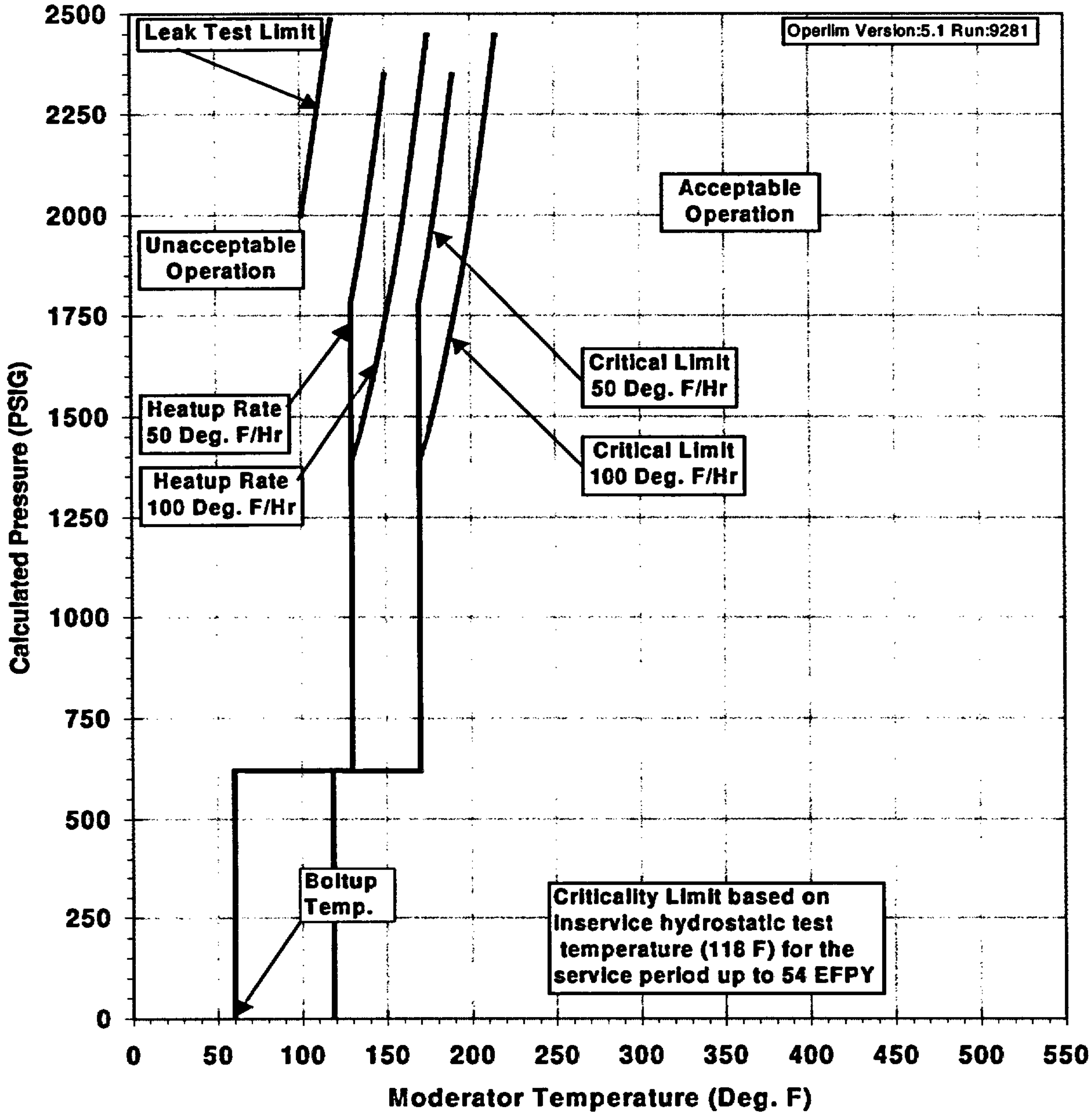


Figure 5.3-2

AP1000 Reactor Coolant System Heatup Limitations (Heatup Rate Up to 50 and 100°F/hour) Representative for the First 54 EFPY (Without Margins for Instrumentation Errors)

From DCD Revision 7, page 5.3-33:

Current Figure 5.3-3

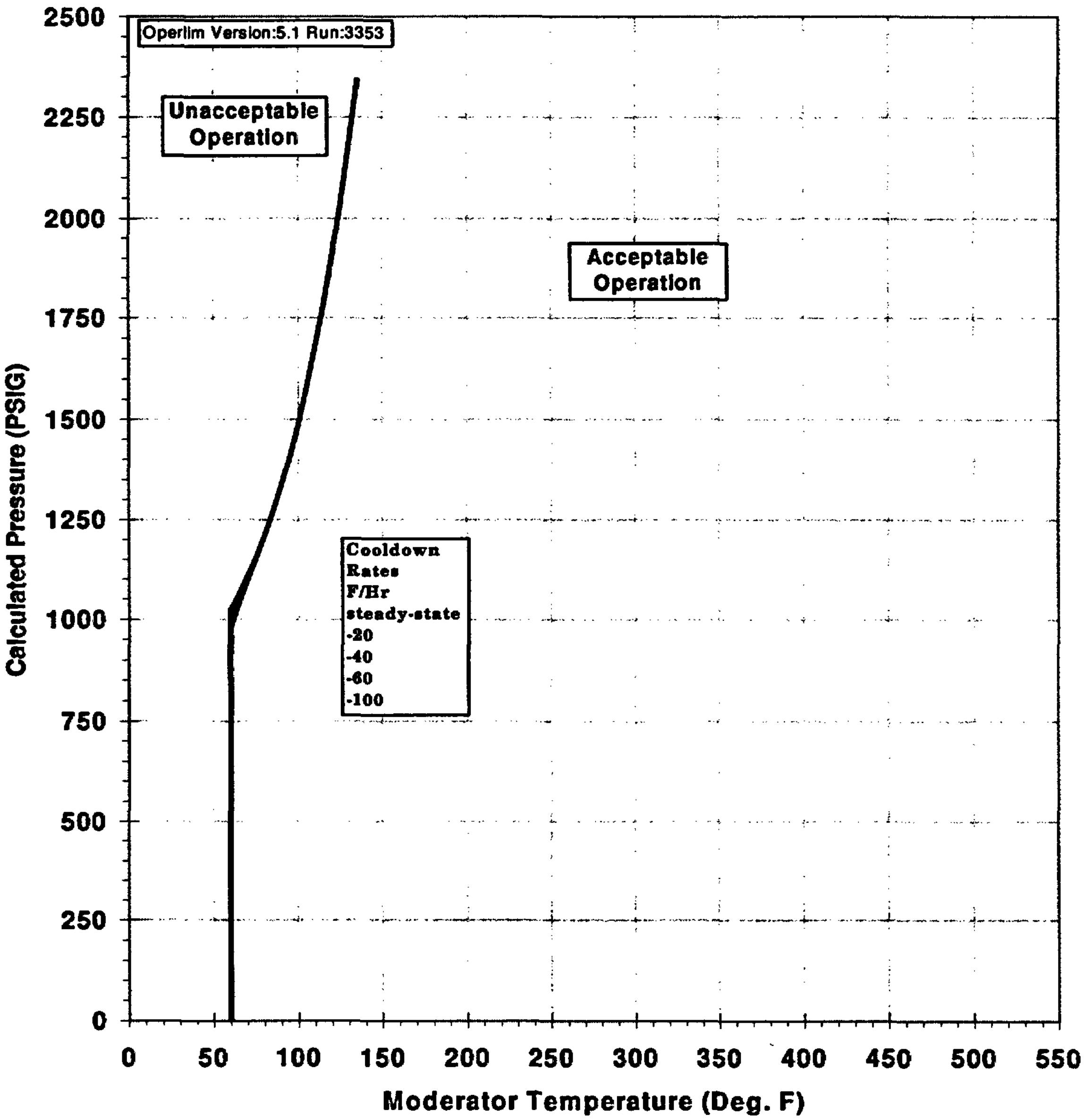


Figure 5.3-3

AP1000 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 50 and 100°F/hour) Representative for the First 54 EFPY (Without Margins for Instrumentation Errors)

Revised Figure 5.3-3

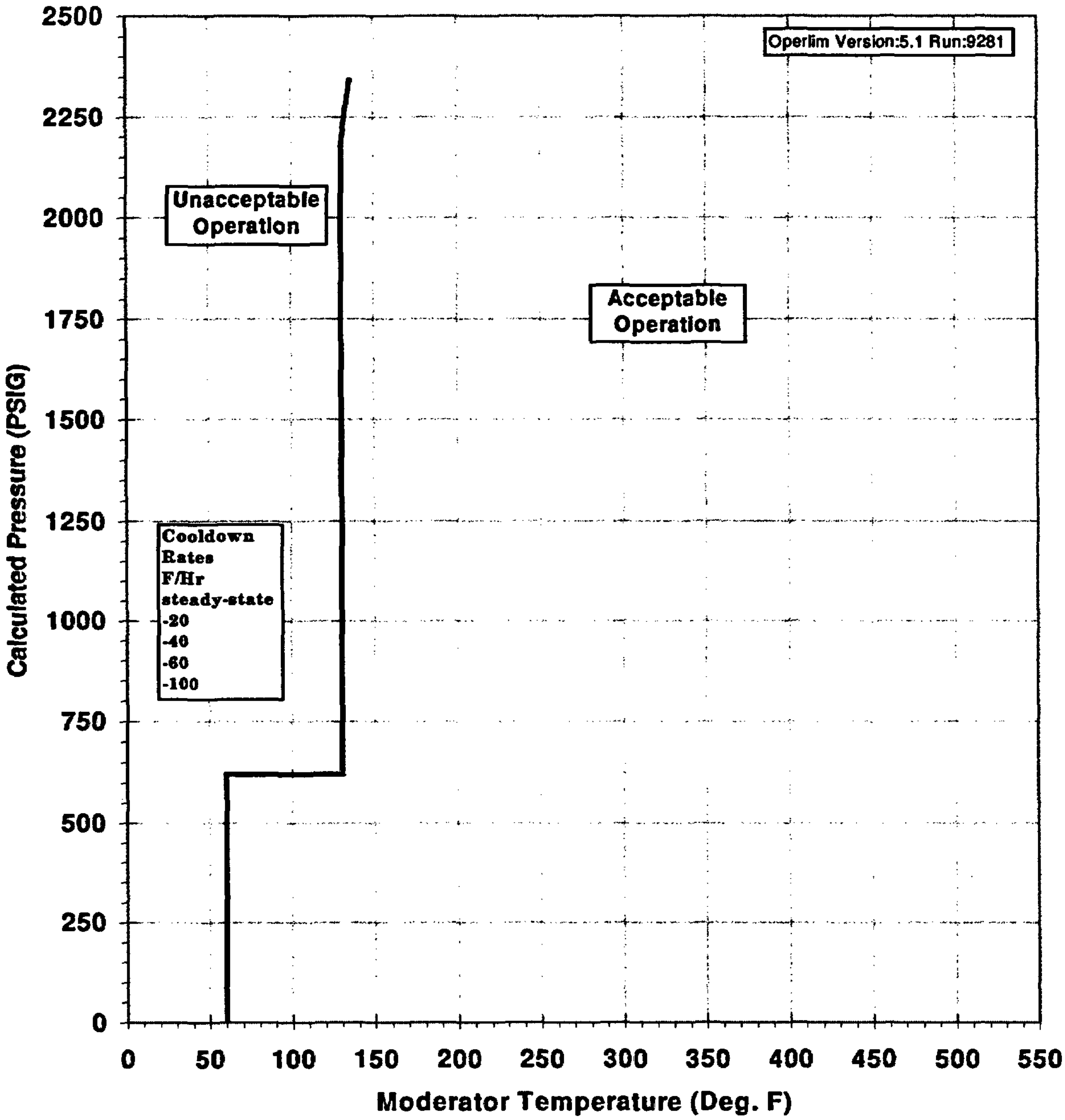


Figure 5.3-3

AP1000 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 50 and 100°F/hour) Representative for the First 54 EFPY (Without Margins for Instrumentation Errors)

From DCD Revision 7, page 5.4-61, Section 5.4.9.3:

The relief valve on the normal residual heat removal system has an accumulation of 10 percent of the set pressure. The set pressure is the lower of the pressure based on the design pressure of the residual heat removal system and the pressure based on the reactor vessel low temperature pressure limit. The pressure limit determined based on the design pressure includes the effect of the pressure rise across the pump. The set pressure in Table 5.4-17 is based on the ~~design pressure of the residual heat removal system~~**reactor vessel low temperature pressure limit**. The lowest permissible set pressure is based on the required net positive suction head for the reactor coolant pump.

From DCD Revision 7, page 5.4-93:

Table 5.4-17

PRESSURIZER SAFETY VALVES - DESIGN PARAMETERS

Number	2
Minimum required relieving capacity per valve (lb/hr)	750,000 at 3% accumulation
Set pressure (psig)	2485 ±25 psi
Design temperature (°F)	680
Fluid	Saturated steam
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	500
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100

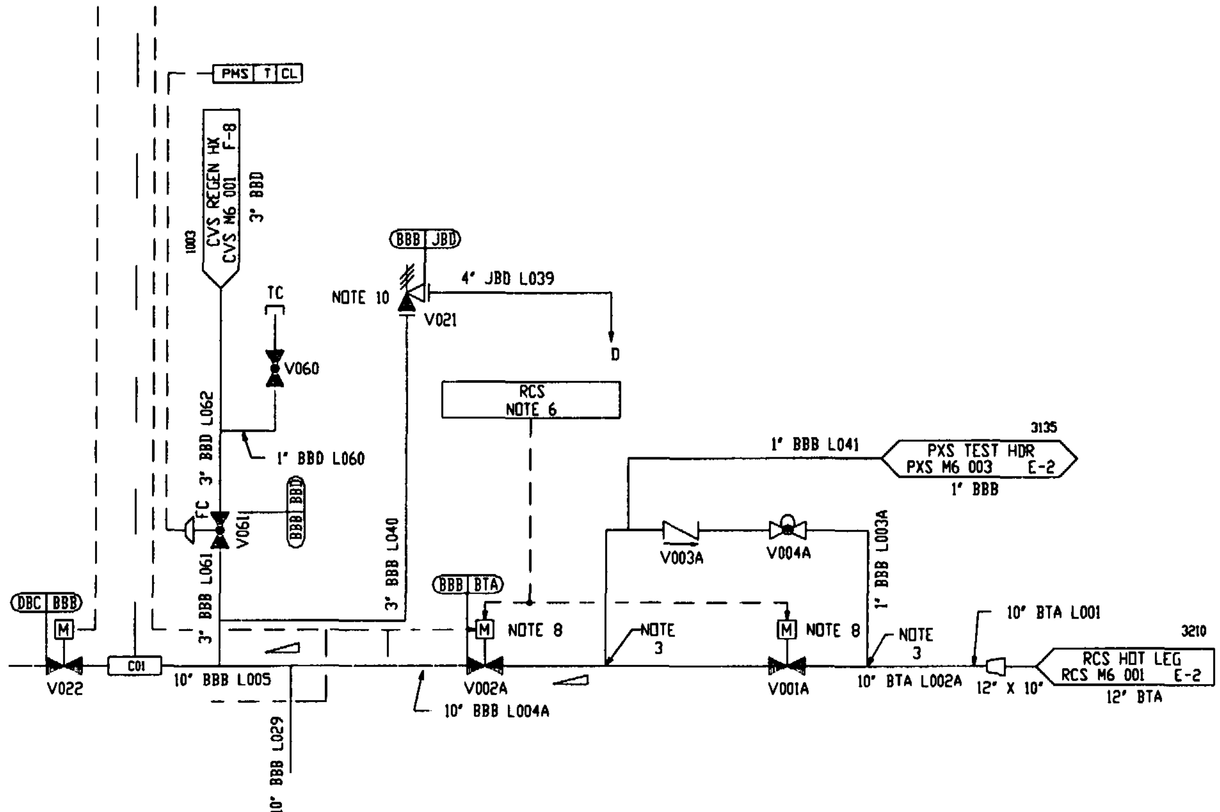
Residual Heat Removal Relief Valve - Design Parameters

Number	1
Nominal relieving capacity per valve, ASME flowrate (gpm)	~~750~~850
Nominal set pressure (psig)	~~636~~500*
Full-open pressure, with accumulation (psig)	~~700~~550*
Design temperature (°F)	400
Fluid	Reactor coolant
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	200
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100

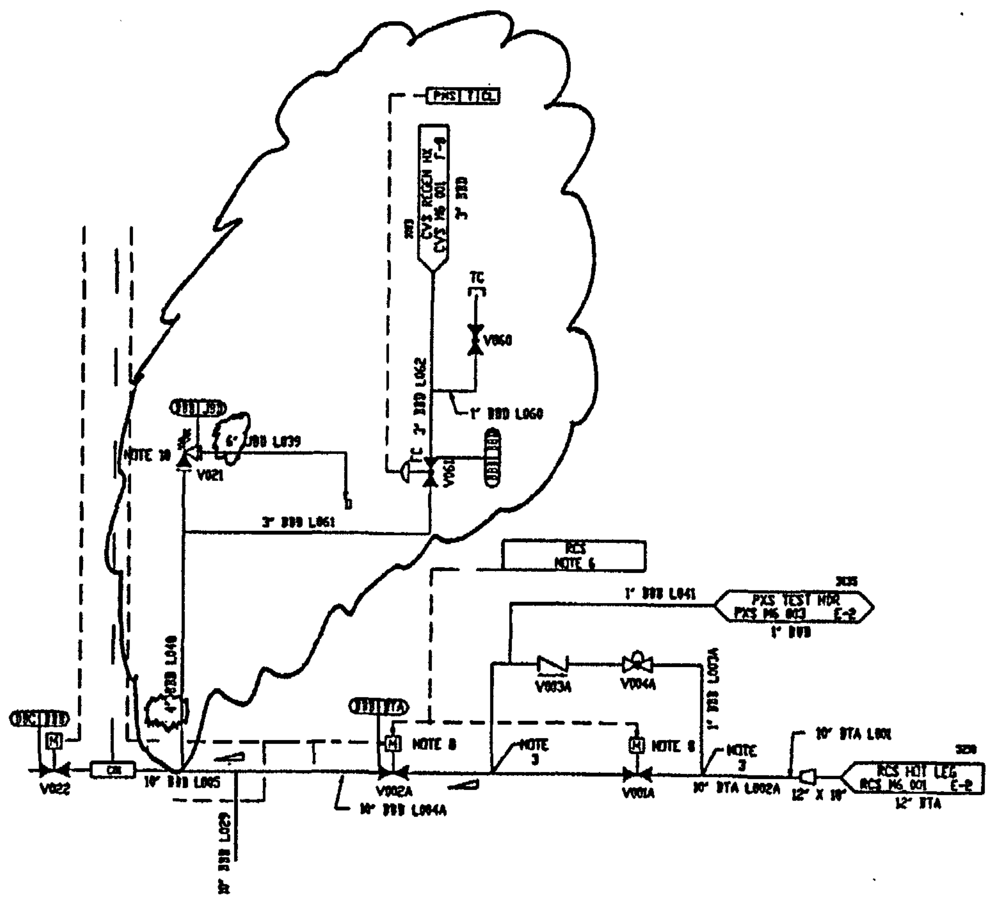
* See text (5.4.9.3) for discussion of set pressure

From DCD Revision 7, page 5.4-107:

Current Figure 5.4-7 Normal Residual Heat Removal System Piping and Instrument Diagram



Revised Figure 5.4-7 Normal Residual Heat Removal System Piping and Instrument Diagram



From DCD Revision 7, page 3.4.14-1, Section 16.1 Technical Specifications, LTOP System:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.14 At least one of the following Overpressure Protection Systems shall be OPERABLE, with the accumulators isolated:

a. The Normal Residual Heat Removal System (RNS) suction relief valve, or

b. The RCS depressurized and an RCS vent of ≥ [~~5.4~~9.3] square inches.

- NOTE -

When the RCS temperature is ≥ 200°F, a reactor coolant pump (RCP) may not be started if the pressurizer level is ≥ 92%.

From DCD Revision 7, page 3.4.14-2, Section 16.1 Technical Specifications, LTOP System:

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. The RNS suction relief valve inoperable.	C.1	Restore the RNS suction relief valve to OPERABLE status.	12 hours
	<u>OR</u>		
	C.2	Depressurize RCS and establish RCS vent of ≥ [~~5.4~~9.3] square inches.	12 hours

From DCD Revision 7, page 3.4.14-3, Section 16.1 Technical Specifications, LTOP System:

	SURVEILLANCE	FREQUENCY
SR 3.4.14.3	- NOTE - Only required to be performed when complying with LCO 3.4.14.b. Verify RCS vent ≥ [~~6.4~~9.3] square inches is open.	12 hours for unlocked-open vent <u>AND</u> 31 days for locked-open vent
SR 3.4.14.4	Verify the lift setting of the RNS suction relief valve.	In accordance with the Inservice Testing Program

From DCD Revision 7, page B 3.4.14-3, Section 16.1 Technical Specifications, Basis 3.4.14, LTOP System:

<u>RNS Suction Relief Valve Performance</u>

Since the RNS suction relief valve does not have a variable P/T lift setpoint, the analysis must show that with chosen setpoint, the relief valve will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the minimum of either the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system. The current analysis shows that up to a temperature of ~~100~~70°F, the mass input transient is limiting, and above this temperature the heat input transient is limiting.

From DCD Revision 7, page B 3.4.14-4, Section 16.1 Technical Specifications, Basis 3.4.14, LTOP System:

<u>RCS Vent Performance</u>

With the RCS depressurized, a vent size of [~~5~~49.3] square inches is capable of mitigating a limiting overpressure transient. The area of the vent is equivalent to the area of the inlet pipe to the RNS suction relief valve so the capacity of the vent is greater than the flow possible with either the mass or heat input transient, while maintaining the RCS pressure less than the minimum of either the maximum pressure on the P/T limit curve or 110 percent of the design pressure of the normal residual heat removal system.

The required vent area may be obtained by opening one ADS Stage 2, 3, or 4 flow path.

The RCS vent size will be reevaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

From DCD Revision 7, page B 3.4.14-4&5, Section 16.1 Technical Specifications, Basis 3.4.14, LTOP System:

The elements of the LCO that provide low temperature overpressure mitigation thr pressure relief are:

a. One OPERABLE RNS suction relief valve; or

   An RNS suction relief valve is OPERABLE for LTOP when both RNS suction isolation valves in one flow path are open, its setpoint is within limits, and testing has proven its ability to open at this setpoint.

LCO (continued)

b. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of ≥ [~~5~~49.3] square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

From DCD Revision 7, page B 3.4.14-7, Section 16.1 Technical Specifications, Basis 3.4.14, LTOP System:

<u>SR 3.4.14.3</u>

The RCS vent of ≥ [~~6.~~49.3] square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context) or

b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position or a removed pressurizer safety valve or open manway also fits this category).

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.14b.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19A.2-7 (Response Revision 2)

Original RAI Number(s): None

Summary of Issue:

The applicant determined the HCLPF values on the basis of the estimated lower bound of qualification test results. When natural frequencies were not known, it was assumed that the equipment natural frequency coincides with the response spectra peak. When equipment frequencies are known and used for comparing the required response spectra (RRS) to the test response spectra (TRS), this information is to be included in the design specification. The applicant has not identified any equipment for which such design specification will be included. Although the applicant appears to have used a conservative approach to obtain the equipment HCLPF value from test results, it is not clear how the use of known natural frequency values for equipment within the standard design scope will be implemented. Since there are many electrical components with HCLPF values at 0.54g and one at 0.53g, electrical components may become critical in determining the Plant HCLPF value. This is Open Item 19A.2-7.

NRC Follow-On Comment:

Replace "will be" by "are" in the last sentence of first paragraph of Westinghouse response.

NRC Comment from 10/6 to 10/10 Audit

Chapter 19 should list the requirements that must be placed in the equipment specification for which the equipment is to be qualified.

Westinghouse Response (Revision ~~1~~2):

The ~~design~~ **equipment** specification is part of the procurement package. The requirements to which the equipment is to be purchased are included in the ~~design~~ **equipment** specification. This includes all those pieces or classes of equipment that have known frequencies that are used to define the HCLPF by comparing the RRS and TRS. These frequencies are included in the ~~design~~ **equipment** specification for the equipment to assure that the dynamic characteristics are the same as those expected. **Chapter 19 is revised to include COL actions related to these specifications.**

Electrical components for non-safety systems are not critical in determining the plant HCLPF value since all SMA sequences are evaluated with loss of offsite power and loss of onsite AC power leading to a station blackout event. With the loss of power it has been shown that the plant design is robust against seismic event sequences each of which contain station blackout coupled with other seismic or random failures.

Design Control Document (DCD) Revision:

~~None~~Revise subsection 19.59.10.5

The Combined License applicant referencing the AP1000 certified design should compare the as-built SSC HCLPFs to those assumed in the AP1000 seismic margin evaluation. Deviations from the HCLPF values or assumptions in the seismic margin evaluation should be evaluated to determine if vulnerabilities have been introduced. **The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include:**

1. **Specific minimum seismic requirements consistent with those used to define the Table 19.55-1 HCLPF values.**

   This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The range of frequency response that is required for the equipment with its structural support is defined.

2. **Hardware enhancements that were determined in previous test programs and/or analysis programs will be implemented.**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **19A.2-9 (Response Revision 2)**

Original RAI Number(s): **None**

Summary of Issue:

Generic Fragility Data

When HCLPF values could not be determined by using one of the methods described above, Westinghouse used generic fragility data. The cases where this approach was used are the following:

- Reactor internals and core assembly that includes fuel
- Control rod drive mechanism (CRDM) and hydraulic drive units
- Reactor coolant pump
- Accumulator tank
- Piping
- Cable trays
- Valves
- Main control room operation and switch stations
- Ceramic insulators
- Battery racks

The generic fragility data came from the Utility Requirements Document which was reviewed by the NRC. Therefore, the use of generic fragility data developed by a joint industry group in the Utility Requirements Document is acceptable. However, the applicant has not indicated what amplification factor, if any, was used to adjust the generic fragility data for the AP1000 configuration. The PCS water flow transmitter, located at Elevation 261' with a HCLPF value of 0.53 g, is likely to have an amplified seismic response. The applicant needs to justify the HCLPF values in the range of 0.53 g and 0.73 g that were obtained from the generic data as shown in the AP1000 PRA Table 55-1, Sheet 3 of 4. This is Open Item 19A.2-9.

NRC Follow-On Comment:

For the generic data used, compare the seismic demand with capacity of component. The NRC asked for all the generic components to be shown in the table that include piping and cable trays.

Westinghouse Response (Revision 1):

No amplification factor was used to adjust generic fragility data for the AP1000 plant. These generic fragility data are considered representative of the anticipated capacity. The Utility Requirements Document data (Reference 19A.2-9-1) are based on plant sites geographically distributed across the central and eastern United States. The generic data can be considered to

provide a measure of the spatial variation of the seismic hazard east of the Rocky Mountains. Rock and four different soil types (EPRI Soil Categories 2 to 5) are considered. Therefore, the seismic input for these generic sites is considered representative to the AP1000 plant whose SSE demand is based on a 0.3g seismic level with a modified Reg. Guide 1.60 spectrum (increased in the higher frequency region around 25 hertz) located on a rock site. The layout and design of the AP1000 plant is similar to the generic plants below the operating deck (El. 135') where the bulk of the safety related equipment is located, with very little outside of the containment. Therefore, the seismic demand defined by frequency content and seismic level will be similar to those associated with the Reference 19A.2-9-1 defined generic plants that are located east of the Rocky Mountains.

A comparison of the generic data is made using the median capacity expressed in terms of spectral acceleration as given in the Utility Requirements Document (Reference 19A.2-9-1) to the maximum AP1000 safe shut down earthquake (SSE) seismic response in Table 19A.2-9-1. It is noted that the ceramic insulators used recognized industry low fragility data and not Reference 19A.2-9-1.

Table 19A.2-9-1 – Comparison of Generic Seismic Response and AP1000 SSE Seismic Response

Description	Frequency Range (Hertz)	Median Capacity Spectral Acceleration (Units: g)	AP1000 Floor Response Spectra (Units: g) (3)	Ratio of Spectral Acceleration
Primary Components		(1)	(2)	(1) / (2)
Reactor Internals and Core Assembly (includes fuel) CIS, El 98'	3-10	3.3	< 1.5	> 2.2
CRDM and Hydraulic Drive Units CIS, E. 98'	5-10	4.7	< 1.5	> 3.1
Reactor Coolant Pump CIS El. 107' 2"	5-10	4.7	≤ 1.5	≥ 3.1
Mechanical Equipment				
Piping CIS, ≤ El 135'	2-10	9.0	≤ 2.5	≥ 3.6
Piping CIS, ≤ El 169'	2-10	9.0	≤ 7.5	≥ 1.2
Piping ASB, ≤ El 135'	2-10	9.0	≤ 3.0	≥ 3.0

Description	Frequency Range (Hertz)	Median Capacity Spectral Acceleration (Units: g)	AP1000 Floor Response Spectra (Units: g) (3)	Ratio of Spectral Acceleration
Piping ASB, ≤ El. 333'	2-10	9.0	< 9	> 1.0
Cable Tray CIS, ≤ El. 135'	5-10	4.7	≤ 2.5	≥ 1.9
Cable Tray CIS, ≤ El. 169'	5-8 (1)	4.7	≤ 4.5	> 1.0
Cable Tray ASB, ≤ El. 135'	5-10	4.7	≤ 3.0	≥ 1.6
Cable Tray ASB, ≤ El. 333'	6-10 (1)	4.7	≤ 4.7	≥ 1.0
Accumulator Tank CIS, El. 98'	5-10	4.7	< 1.5	> 3.1
Valves CIS, ≤ El 169'	> 20	9.0	≤ 5.7	≥ 1.6
Valves ASB, ≤ El 333'	> 20	9.0	≤ 2.5	≥ 3.6
Electrical Equipment				
Battery Racks ASB, El < 81.5'	8-12	7.0	< 1.5	> 4.6
Ceramic Insulators		0.3	(2)	
MCR Support Operation Work Station & MCR Switch Station ASB, 116.5'	4-12	5.9	< 2.25	> 2.6

Notes to Table 19A.2-9-1:

(1) Small change in generic frequency range recognizing that the cable tray systems are designed away from spectral peaks that are large.
(2) The Capacity is less than review level earthquake, 0.5g. Failure is assumed in PRA evaluation.
(3) Based on 5% equipment damping.

As seen from Table 19A.2-9-1 no amplification factor is needed to adjust generic fragility data for the AP1000 plant. These generic fragility data are considered representative of the anticipated capacity.

It is noted that in chapter 55, Section 55.2.1, Westinghouse identified the following COL actions to confirm the seismic margin evaluation that includes generic fragility data:

> As part of a COL action, a qualification seismic review of the design will be performed with the purpose of identifying vulnerabilities and confirming the basis of the seismic margin evaluation. For each plant, a verification walkdown will be performed with the purpose of identifying differences in the as built from design and ensuring vulnerabilities were not created.

References

19A.2-9-1 ALWR URD, Volume III, ALWR Passive Plant, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules, Revisions 5 & 6.

Design Control Document (DCD) Revision:

None

PRA Revision:

None